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Abstract

A Sodium Cooled Power Reactor Experiment Employing Zirconium Hydride Moderator

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The characteristics and some details of the design for a 20 MWe sodium cooled reactor experiment are to be described. The heterogeneous core will be fuelled by steel-canned, slightly enriched UO_2, using zirconium hydride as the moderator. The heat transfer system has been designed for a maximum sodium temperature of 560°C to generate steam of 510°C at 85 atm pressure. Essentially all system parts in contact with high temperature sodium are to be constructed of a low alloy ferritic steel that does not decarburize on hot sodium.

The design has already been supported by an extensive experimental development effort over a period of several years. The program comprises a number of reactor-physics experiments (neutron spectra measurements, pulsed subcritical and critical experiments), the development and the investigation of zirconium hydride moderator elements, the testing and the evaluation of a low-alloy ferritic material for high-temperature sodium applications, systems- and core components development by means of various sodium loops including a 5-thermal-Megawatt Sodium-Test-Facility (permitting the operation of sodium heated steam generators up to 540°C and 160 atm pressure), investigation of steam tube ruptures in sodium heated steam generators etc. Some results of the experimental program are to be given.

Finally a comprehensive survey of the economical and technical aspects of the reactor type considered will be outlined.